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This document consists
of 10 pages No. 6
of 29 Series A

Doc. No. DC 61-1-1

AIRCRAFT NUCLEAR PROPULSION DEPARTMENT

ChemRisk Document No. 2753

SUMMARY OF FISSION PRODUCT RELEASE FROM ORR TESTS

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December 29, 1960 DECLASSIFIED By Authority Of: AEC May 1970

This document has been approved for release
to the public by:

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Serial No. 16464

GENERAL ELECTRIC

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SUMMARY OF FISSION PRODUCT RELEASE FROM ORR TESTS

This document summarizes the fission product release behavior from a series of ORR tests. Its purpose is to present data as they presently exist with a few comments regarding interpretation, for use in internal discussions regarding fission product release from the ACT type of fuel element. The writer would welcome any discussion regarding details or interpretation.

The results from six ORR tests are summarized in the attached tables and figures. The first test (ORC-1) used BF-116 tubes coated on both sides with Al_2O_3 and the remaining tests used BF-116 tubes coated with ZrO_2 on the bore only.

All of the ORR tests except ORC-3 have achieved a burnup that is in excess of that expected in the ACT reactor, 2.5×10^{19} fissions/cc. Tests ORC-3 and -4 were run in dry air instead of the normal wet air of $25^{o}F$ D.P. Only tests ORC-1 and -2 have achieved 1000 hrs., ORC-5 achieved ~920 hours of irradiation, the other tests were terminated at shorter periods due to limited operation of the ORR reactor. ORC-6 is currently on test.

ORC-2

With the exception of ORC-1, all tests showed an increase in the iodine-131 release above 0.1% at some time during the test. ORC-2 has shown the greatest iodine-131 release, reaching values of the order of 6% at the 1000 hr. period. The increasing release values during the test were attributed to corrosion on the backside as indicated by post irradiation examination. The increase in the release seemed to rise at an accelerating rate which was not in agreement with predictions. Some of the increase may have been due to higher operating temperatures than the thermocouple indicated.

The Sr-89 release also increased steadily during the test which is unusual even for tests run at higher temperatures. Backside corrosion however would be responsible for such an increase.

ORC-3 and -4

These tests were run in dry air to eliminate the extraneous release of fission products from backside corrosion. In ORC-3 the iodine-131 release remained below 0.1% for 400 hours but the last two traps between 400-460 hours indicated release values of 0.15 - 0.18%. The Sr-89 release also was below the normal upper limit of 0.02% except for the last two traps. The trap data between 80 and 260 hrs. of testing were not reliable since the trap system developed a leak. In ORC-4 the I^{131} release remained below 0.1% for over 400 hours of testing but in the last two hundred hours showed releases which were between 0.1 and 0.3%. The Sr^{89} values, as in ORC-3, remained below 0.02% throughout the test except for two transient traps late in the test period. Comparison of the total iodine-131 release values for ORC-2 and ORC-4 at 600 hrs. point of testing shows that the release of iodine was higher in wet air by a factor of three. Furthermore, the Sr-89 release in ORC-2 was higher than the release in ORC-3 and -4 by a factor of five indicating that corrosion on the backside contributed considerably to the high release values in ORC-2. The sudden rise in release in ORC-3 and -4 late in the test period is puzzling. Such effects as burn-up, corrosion, and diffusion would show only gradual increases in the release with irradiation. Recently laboratory tests have indicated that the thermocouples used in the ORR tests show a tendency to decalibrate. Also flow measurements on MTR assemblies indicated that a ~~large~~ percentage (5-40) of the total flow could pass through the backside channels of the assembly possibly cooling the thermocouple below the true fuel element temperature.

As a result of these factors, two features were incorporated into subsequent tests.

1) A wall temperature was calculated from a thermodynamic analysis of the operating conditions during the test and was compared with thermocouple readings.

2) The tube bundle was tightly wrapped and squeezed together by Thermoflex insulation under the action of two ceramic shells to provide a minimum of space for flow along backside channels.

Item (2) was incorporated in tests ORC-4 and -5. Item (1) was carried out during ORC-5.

ORC-5

In ORC-5 the iodine-131 release remained below 0.1% and Sr-89 below 0.02% in the first 80 hours of the test. A leak developed in the trap system depriving any further information until 390 hours of testing. During this time the temperature calculated from thermodynamic data showed a steady rise while the controlling thermocouple indicated no change.

At 450 hours the reactor was shut down for refueling. Approximately five hours after start-up, the line activity rose sharply from a normal 4-6 r/hr. to 20 r/hr or about 5 times higher than what was noted previously. The calculated temperature during this interval was about 2900°F or near the BeO-Y_2O_3 eutectic temperature where large release of fission products might be expected. Unfortunately no traps were operating at the time. The test hereafter was controlled by thermocouple temperatures at the entrance end. The calculated temperature and line activity were reduced sharply as a result. Subsequent iodine-131 release values averaged 0.3% higher than the normal average of 0.08%. Sr-89 release values were also higher by a factor of two. It may be possible that some permanent damage to the fuel element such as fuel loss occurred as a result of the temperature excursion at 450 hrs., resulting in the net increase in both Sr^{89} and I^{131} values.

Since neither of these two isotopes increased gradually before or after the temperature excursion, it is evident that the effects of backside corrosion on release are sharply reduced below the values obtained in ORC-2 where there was no attempt to minimize backside flow. Visual examination of ORC-5 will determine further if this observation is valid.

ORC-6

Early results from test ORC-6 which is being run at $\sim$2000°F to determine thermocouple integrity indicate that both iodine-131 and Sr-89 isotopes are lower than the respective average values of 0.08% and 0.02% observed from tests at higher temperatures. If this condition remains over a significant test period, average release estimated for the ACT (DCL 60-11-57) may be somewhat conservative since a significant fraction ($\sim$50%) of the fuel elements will be below this 2000°F.

These series of tests indicate so far the following conclusions:

1) Temperature measurement with the present thermocouples is extremely unreliable as evidenced by decalibration in laboratory experiments and is the primary deterrent to making firm conclusions about fission product release after extensive burn-up and heat treatment.

2) Aside from the temperature measurement problem there is indirect evidence that fission product release from backside corrosion can be drastically diminished by restricting backside flow.

3) Effect of burn-up at least to 2.5×10^{19} f/cc on fission product release is negligible.

SUMMARY OF CERAMIC TESTS IN ORR F-2 FACILITY

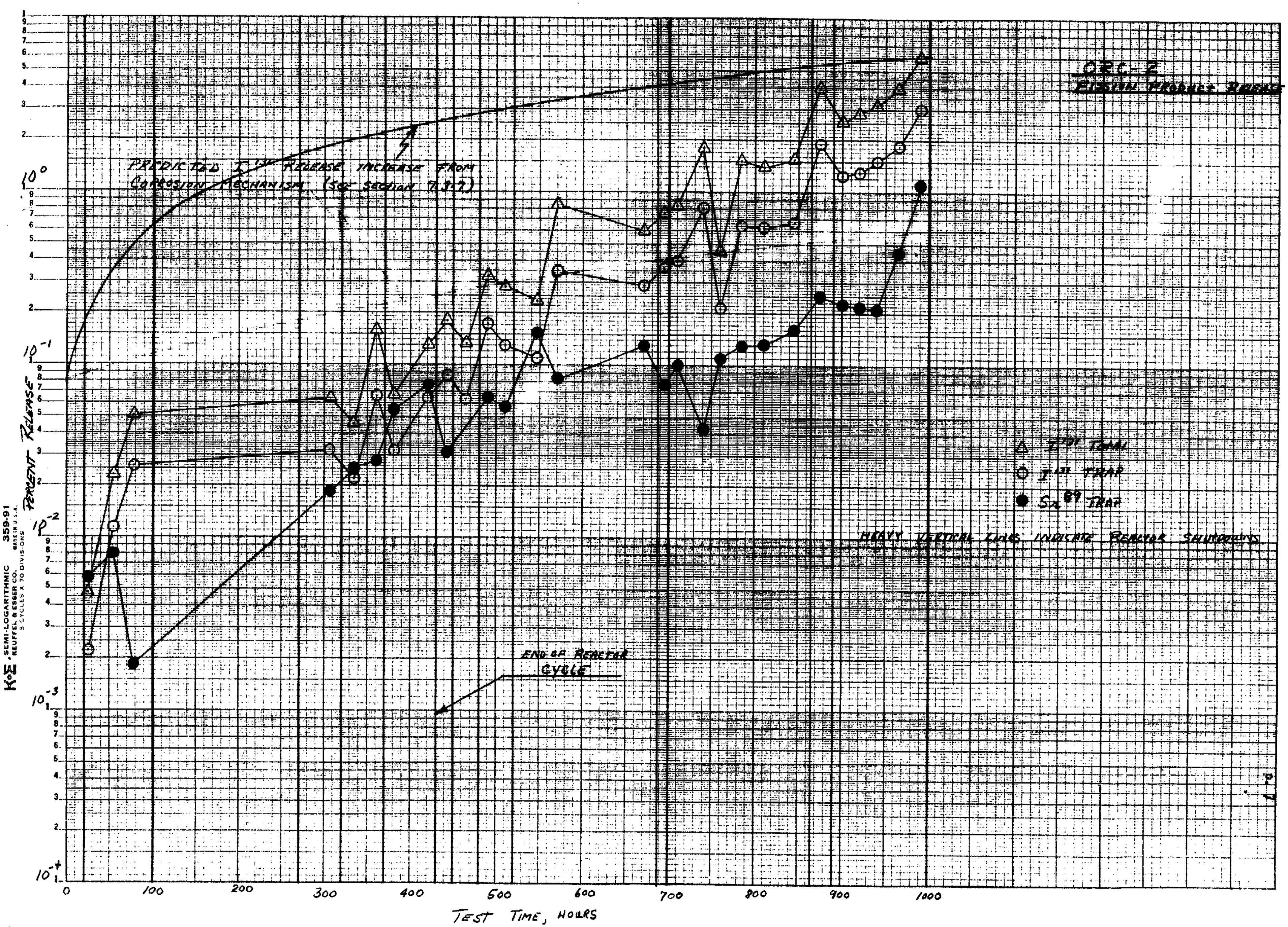
Test Number	Nominal Matrix Composition	Clad[a]	Average Maximum Indicated Temp., °F of Outer Dia.	D.P. °F	Test Time, hr.	Average Fissions[b] per $cm^3 \times 10^{-19}$	Maximum Calculated Thermal Stress: Residual, Outer[c] Fiber σ_o, psi	Maximum Calculated Thermal Stress: Operating, Inner[d] Fiber σ_i, psi
ORC-1	8BF-116	Al_2O_3	2500	+25	1000.2	5.42	19,500	8,660
ORC-2	8BF-116	ZrO_2	2600	+25	1000	4.70	10,700	4,340
ORC-3	6BF-116	ZrO_2	2600	-30	467.9	1.38	15,900	9,700
ORC-4	6BF-116	ZrO_2	2600	-50	599.6	3.40	20,700	10,800
ORC-5	6BF-116	ZrO_2	2600	+25	~920	5.1	(20,700)	(10,800)
ORC-6	6BF-116	ZrO_2	2000	-50	on test			

a) The ORC-1 insert was clad on both inner and outer surfaces; all other inserts were clad on the bore only.

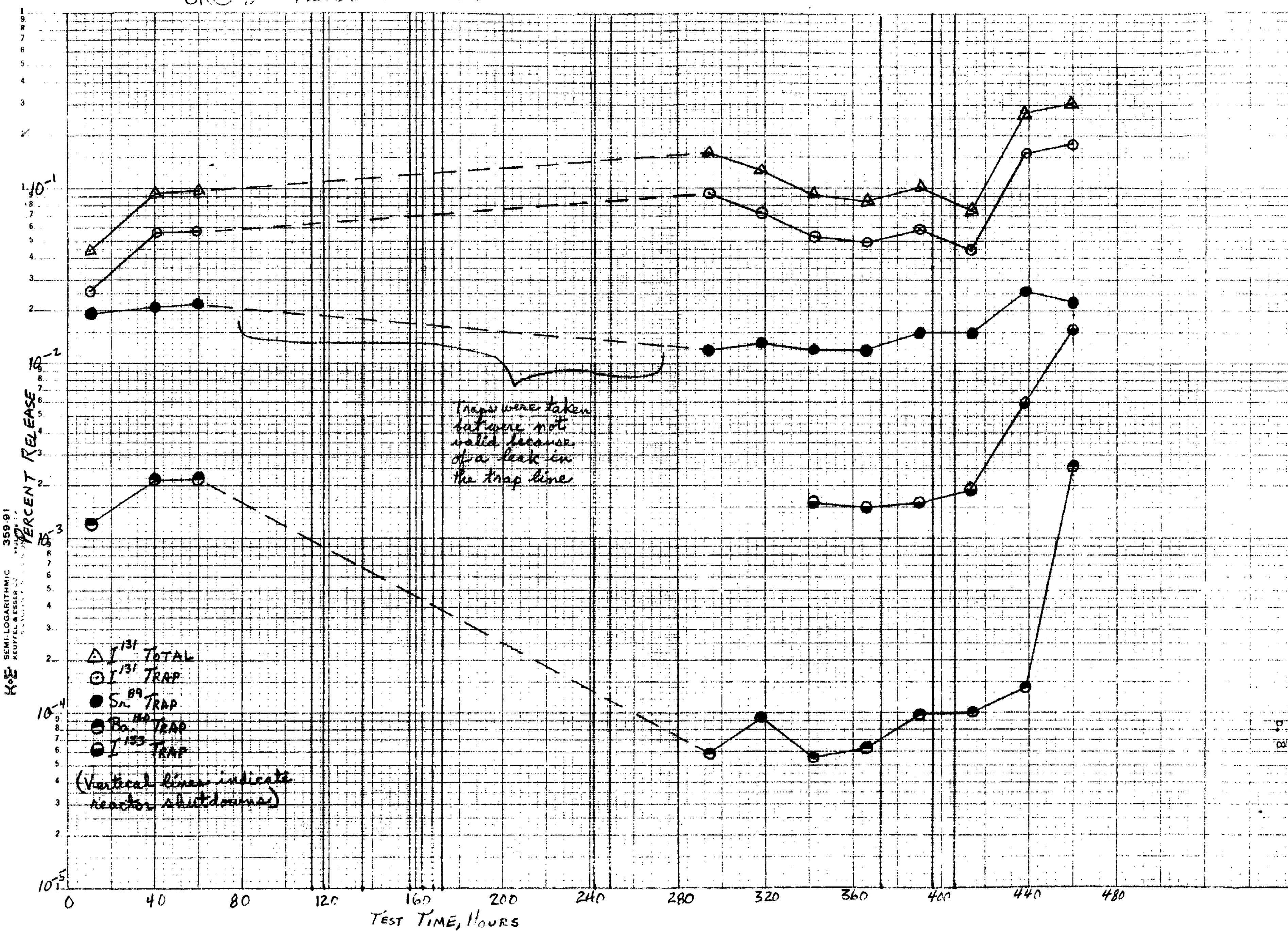
b) The fission density for ORC-1 and ORC-2 was determined by dividing the total number of fissions (determined by post-test radiochemical analyses and gamma scannings) by the tube volume. Since radiochemical analyses have not yet been obtained on the three other tests, the fission densities were calculated by multiplying the fission density on ORC-1 by ratios of the flux-times, fuel densities, and average perturbed-flux values corrected for actual operating reactor-power levels. To convert them to fission densities based on matrix volume alone, each of the above values is divided by the values of the volume fraction of the matrix given in subsequent data sheets.

c) Calculated using a corrected maximum q''' (Btu/sec-in.^{3}) prior to a shutdown (includes hexagonal and end effects).

d) Calculated using a corrected maximum operating q''' (includes hexagonal and end effects) at operating temperature.


ORC-2
FISSION PRODUCT RELEASE
PREDICTED I131 RELEASE INCREASE FROM
CORROSION MECHANISM (SEE SECTION 7.3.7)
I131 TOTAL
I131 TRAP
Sr89 TRAP
HEAVY VERTICAL LINES INDICATE REACTOR SHUTDOWNS
END OF REACTOR
CYCLE
PERCENT RELEASE
TEST TIME, HOURS
0
100
200
300
400
500
600
700
800
900
1000
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
K+E SEMI-LOGARITHMIC 359-91
KEUFFEL & ESSER CO. MADE IN U.S.A.
5 CYCLES X 70 DIVISIONS
p. 7

ORC-3 Fission Product Release



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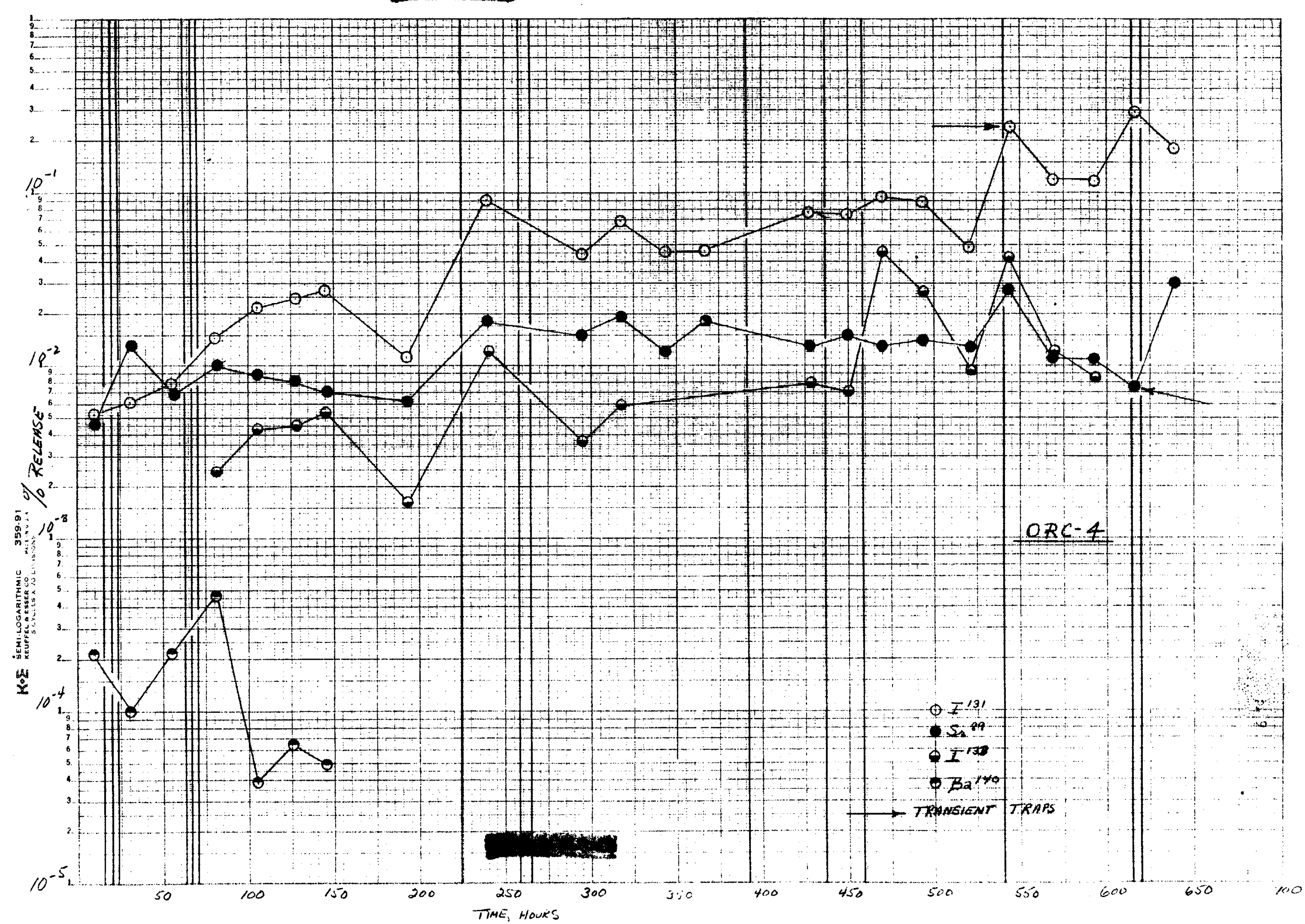

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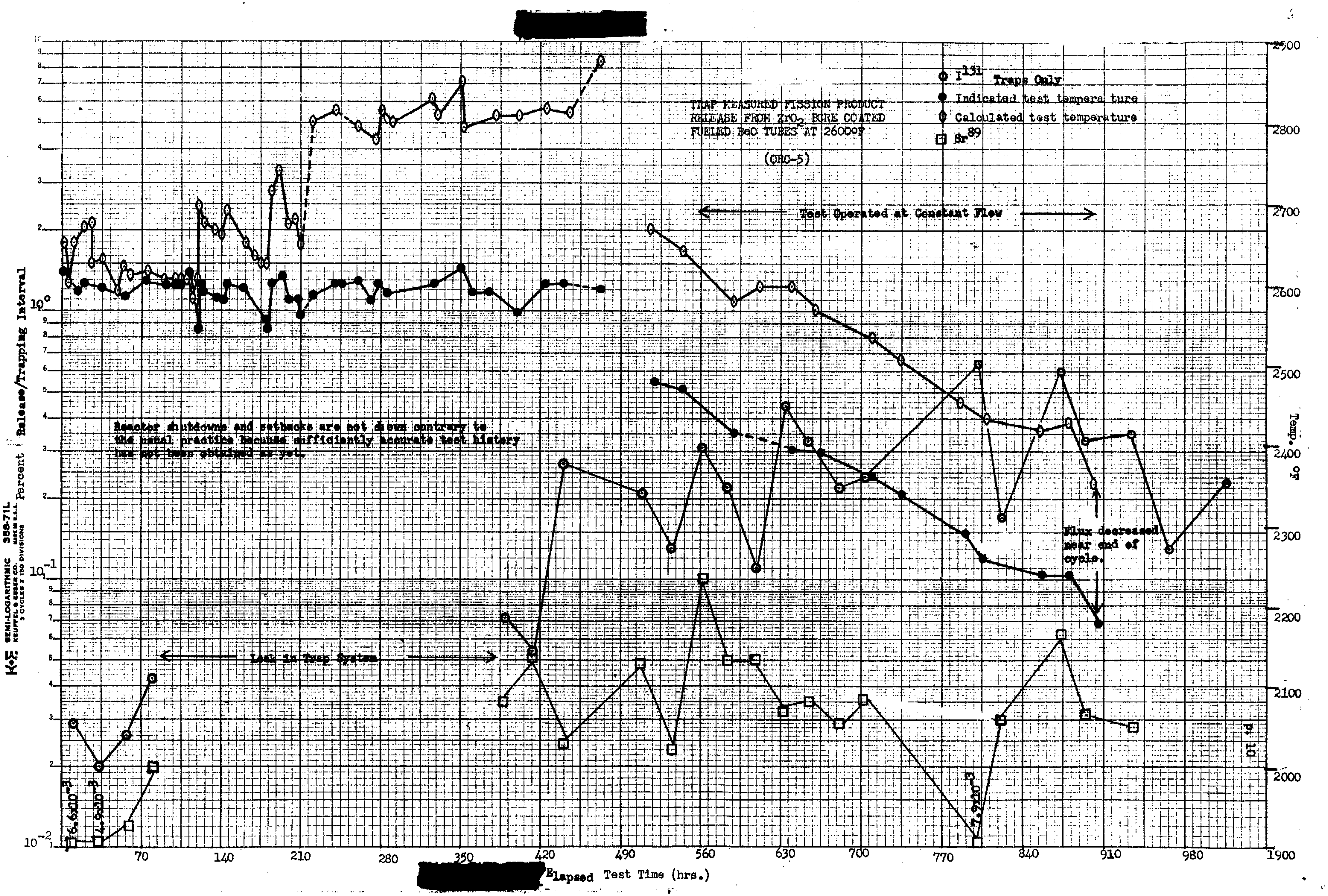
[redacted]
TRAP MEASURED FISSION PRODUCT
RELEASE FROM ZrO2 BORE COATED
FUELED BeO TUBES AT 2600°F
(ORC-5)
I131 Traps Only
Indicated test tempera ture
Calculated test temperature
Sr89
Test Operated at Constant Flow
Reactor shutdowns and setbacks are not shown contrary to
the usual practice because sufficiently accurate test history
has not been obtained as yet.
Leak in Trap System
Flux decreased
near end of
cycle.
6.6x10-3
4.9x10-3
7.9x10-3
Percent Release/Trapping Interval
Temp. °F
[redacted] Elapsed Test Time (hrs.)
p. 10
K+E SEMI-LOGARITHMIC 358-71L
KEUFFEL & ESSER CO.
3 CYCLES X 150 DIVISIONS